ACCESSION #: 9602210154

LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK STEAM ELECTRIC

STATION UNIT 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP DUE TO MAIN STEAM LINE LOW PRESSURE SAFETY

INJECTION SIGNAL

EVENT DATE: 01/17/96 LER #: 96-001-00 REPORT DATE: 02/16/96

OTHER FACILITIES INVOLVED: CPSES UNIT 1 DOCKET NO: 05000445

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

NAME: RALPH FLORES, SYSTEM ENGINEERING

MANAGER TELEPHONE: (817) 897-5590

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On January 17, 1996, at approximately 8:01 a.m.(CST), Unit 1 inverter IV1PC2 output breaker tripped on over current. This resulted in the loss of power to distribution panel 1PC2. At approximately 8:08 a.m., the loss of power to the panel (1PC2), which armed the steam dumps, required the Auxiliary Operator (utility, non-licensed) to transfer power to a separate power source. TU Electric believes that this subsequent transfer action to

restore power initiated a spike (high) on the N-16/T sub avg instrumentation creating a temperature error signal to the Steam Dumps. The Steam Dump valves opened resulting in the generation of Main Steam line pressure - low Safety Injection. A reactor trip signal was initiated as a result of the SI signal. ECCS injection was terminated in accordance with the emergency operating procedures. TU Electric believes that the event was caused due to a large T sub avg - T sub ref error signal generated following the restoration of power to the distribution panel. Plant procedures have been revised to prevent repetition of this type of event.

TEXT PAGE 2 OF 7

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in a manual or automatic actuation of any Engineered Safety Features (ESF) including the Reactor Protection System (RPS).

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On January 17, 1996, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, and operating at 100 percent power.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

There were no inoperable structures, systems or components that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On January 17, 1996, at approximately 8:01 a.m., Unit 1 inverter IV1PC2 (EIIS:(INVT)(EF)) output breaker tripped on over current. This resulted in the loss of power to distribution panel 1PC2 (EIIS:(BU)(EF). Several annunciators were received as the following equipment was lost: Letdown flow, Inverter IV1PC2, Automatic Feedwater Control on Steam Generators (EIIS:(SG)(SB) 2 & 3, and the Plant Computer and channel II 7300 process instrument cabinet.

The Reactor Operator (RO) (utility, licensed) took manual control of Feedwater regulating valve flow to the number 2 and 3 steam generators to stabilize levels. The Unit Supervisor (utility, licensed) noted a loss of distribution panel 1PC2 as indicated by the Trip Status Light Boards(TSLB). All channel II TSLBs were lit. The plant computer indicates that the plant computer lost functionality at

approximately 8:01 a.m. due to overload of input data associated with the 1PC2 transient.

Operators entered Comanche Peak abnormal operating procedure for loss of an instrument bus, to recover from the loss of the inverter (1PC2). The Unit Supervisor informed the RO of his intention to reenergize the 1PC2 panel from the alternate power supply. He cautioned operators that perturbations may be experienced on instrumentation that could cause activations or control fluctuations.

At approximately 8:08 a.m., January 17, 1996, Unit 1 power supply for distribution panel (1PC2) was switched to the alternate power source causing a spike which resulted in a spike (high) on the Nuclear Instrument N-16/T sub avg creating an error signal to the Steam Dumps. The Steam Dump valves (EIIS:(RV)(SB)) opened and generated rate compensated low Steam line Pressure and Steam line isolation signals resulting in a Safety Injection(SI) signal on a Main Steam (EIIS:(SB)) Line low pressure signal and reactor trip. Injection was terminated following the SI in accordance with the emergency operating procedures. The plant was stabilized in mode 3.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR

PROCEDURAL ERROR

At approximately 8:01 a.m, several annunciators were received as the following equipment was lost: Letdown flow, Inverter IV1PC2, Automatic Feedwater Control on Steam Generators 2 & 3, and the Plant Computer and channel II 7300 process instrument cabinet.

At approximately 8:08 a.m the Control Room received a Main Steam line Pressure Low - SI actuation signal.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

An over current occurred in the 118 VAC distribution system of the 118 VAC Distribution Panel 1PC2. The over current relay latched and energized a 20 second time delay relay that tripped the output circuit breaker at Inverter IV1PC2.

In order to restore power to distribution panel 1PC2, the AO transferred power to a separate power source. TU Electric believes this transfer initiated a spike (high) on the N-16/T sub avg instrumentation creating a temperature error signal to the Steam Dumps. The Steam Dump valves opened resulting in the generation of low steam line pressure and steam line isolation signals resulting in a Safety Injection signal on a Main Steam Line low pressure signal and reactor trip.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

An over current occurred in the 118 VAC distribution system of the 118 VAC Distribution Panel 1PC2. The over current relay latched and energized a 20 second time delay relay that tripped the output circuit breaker at Inverter IV1PC2.

The loss of the inverter caused the impulse chamber pressure signal from channel PT-506 to fail low, which energized the loss of load bistable PB-506c to unblocking (arming) the steam dumps(as if an actual decrease in turbine impulse pressure had occurred). When distribution panel 1PC2 was reenergized it

initiated a spike (high) on the N-16/T sub avg creating an

error signal to the steam dumps causing them to trip open. If

a safety injection had not occurred first, the steam dumps

would have shut automatically when T sub ave reached the Lo-Lo

setpoint.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF

COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - No failures of components with multiple

functions have been identified.

D. FAILED COMPONENT INFORMATION

Not applicable- Preliminary results of TU Electric's Task Team

(which was organized to investigate the subject event) does not

indicate a component failure issue. The issues with respect to

Inverters will be further discussed in CPSES LER 445/96-002-00.

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

ECCS Essential Ventilation Systems (EIIS:(VA)) (EIIS:(VF))

(EIIS:(VL)) Containment Spray Pumps (EIIS:(P)(BE)) Component

Cooling Water System (EIIS:(CC)) Motor Driven Auxiliary

Feedwater Pumps (EIIS:(P)(BA)) Control Room Heating, Ventilation and Air Conditioning System switched to Emergency Recirculation (EIIS:(VI)) Diesel Generators (EIIS:(EK))

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not Applicable- No safety system trains were inoperable during this event.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The inadvertent actuation of the Steam Dump System is considered to be an ANS Condition II event and is evaluated in the Final Safety Analysis Report (FSAR), Section 15.1.4. The relevant event acceptance criterion for this event is that the calculated minimum departure from nucleate

boiling ration (DNBR) should not fall below the limit value.

In the evaluation presented in FSAR Section 15.1.4, it is concluded that the effects on the core are bounded in severity by the analysis of the main steamline break event presented in FSAR Section 15.1.5. Even though the steamline break event is an ANS Condition IV event, compliance with the Condition II event acceptance criteria is demonstrated. Thus, it is concluded that the relevant event acceptance criteria for the actual event would have been satisfied.

An analysis of an inadvertent ECCS actuation is presented in FSAR Section 15.5.1. This event is also an ANS Condition II event. This event is analyzed to demonstrate that the available time for operator actions to secure the ECCS is sufficient to prevent completely filling the pressurizer with liquid. This requirement is imposed to ensure that the event does not progress to a more serious event (i.e., an ANS Condition III or IV event). In the actual event, the time required to secure the ECCS was greater than assumed in the analysis of Section 15.5.1. However, given the additional activities required by the initiating event (the steam dump actuation), the response time is considered appropriate. Furthermore, the pressurizer water level during the event did not exceed approximately 82% span; thus, the relevant event acceptance criterion was satisfied.

Based on this analysis, it can be concluded that the event had no impact on the health and safety of the public.

IV. CAUSE OF THE EVENT

This January 17, 1995 event was initiated by the loss and subsequent restoration of power to the protection bus 1PC2, which was caused due to the tripping open of the inverter IV1PC2 output breaker. (The cause for the inverter loss and corresponding corrective actions will be addressed in CPSES LER 445/96-002-00). Among the channels affected by the loss of bus 1PC2 were 1-PT-0506, which generated the C-7 (loss of load interlock) steam dump arming signal, and the loop 2 T sub avg, When the operators reenergized 1PC2 from its alternate power supply, the loop 2 T sub ave spiked high and created a +40 degrees F

(approximate) auctioneered high T sub avg - T sub ref mismatch, which in combination with existing steam dump arming signal, tripped open the steam dump valves. The sudden increase of steam flow created a rate compensated main steamline low pressure steamline isolation, Safety Injection and a subsequent plant trip.

TU Electric believes that the abnormal operating procedure developed for the loss of protection or instrument bus did not anticipate N-16 spike before clearing the C-7 interlock.

V. CORRECTIVE ACTIONS

Immediate actions taken were to stabilize the plat to Mode 3. TU Electric management initiated a Task Team investigate the event and develop corrective actions for the issues regarding the Inverters.

A lessons learned has been developed to ensure that plant operators are cognizant of this event. A procedure change has been initiated which should preclude recurrence of the inadvertent steam dump actuation

VI. PREVIOUS SIMILAR EVENTS

There have been no other previous events, which had similar causes that resulted in a Safety Injection actuation signal.

VII. ADDITIONAL INFORMATION

The issues/circumstances regarding the Inverters will be submitted

via CPSES LER 445/96-002-00.

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Log # TXX-96044

File # 10200

TUELECTRIC Ref. # 10CFR50.73(a)(2)(iv)

C. Lance Terry

Group Vice President

February 12, 1996

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, DC 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)-UNIT 1

DOCKET NOS. 50-445

MANUAL OR AUTOMATIC ACTUATION OF ENGINEERED SAFETY FEATURES

LICENSEE EVENT REPORT 445/96-001-00

Gentlemen:

Enclosed is Licensee Event Report (LER) 96-001-00 for Comanche Peak Steam

Electric Station Unit 1, "Reactor Trip due to Main Steam Line Low

Pressure Safety Injection Signal".

Sincerely,

C. L. Terry

By:

M. R. Blevins

Plant Manager

OB:ob

Enclosure

cc: Mr. L. J. Callan, Region IV

Mr. W. D. Johnson, Region IV

Resident Inspectors, CPSES

P. O. Box 1002 Glen Rose, Texas 76043

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